U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION DIVISION OF LICENSE RENEWAL
AGING MANAGEMENT AUDIT REPORT

Docket Nos: 50-247 and 50-286

License Nos: DPR-26, DPR-64

Licensee: Entergy Nuclear Operations

Facility: Indian Point Nuclear Generating Station Units 2 and 3

Location: Westinghouse Twinbrook Office Facility, Rockville, MD

Dates: April 24-25, 2013

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Background

By letter dated April 23, 2007, Entergy Nuclear Operations, Inc. (Entergy, the applicant) applied for renewal of the Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and IP3), collectively known as Indian Point Energy Center (IPEC) operating licenses. The Vessel & Internals Branch (EVIB) staff is reviewing the Indian Point Nuclear Generating Unit Nos. 2 and 3 Reactor Vessel Internals Program provided in Amendment 9 to the IPEC License Renewal Application (LRA), submitted via letter dated July 14, 2010. In conjunction with the review of the reactor vessel internals (RVI) Program, EVIB is reviewing the "Indian Point Energy Center Reactor Vessel Internals (RVI) Inspection Plan," submitted via letter dated September 28, 2011, as supplemented by letter dated February 17, 2012. The RVI Inspection Plan, as supplemented, was intended to be consistent with the U. S. Nuclear Regulatory Commission (NRC)-approved Electric Power Research Institute topical report "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines" (MRP-227-A), and to partially fulfill Commitment #30 of the IPEC LRA. In a letter dated September 28, 2012, the applicant

ENCLOSURE 2

provided responses to staff requests for additional information (RAIs) and stated, for certain RAI responses, that plant-specific details are proprietary and if the NRC requires additional details, the calculation(s) would be made available for the NRC to review the supporting technical basis.

Regulatory Bases

License renewal requirements are specified in Title 10 of the *Code of Federal Regulations* (10 CFR), Part 54 (10 CFR Part 54), "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." Guidance is provided in NUREG-1800, Revision 2, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR), dated December 2010, and in NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," dated December 2010.

Regulatory Audit Scope and Methodology

The scope of this audit was to review the plant-specific, proprietary information associated with RVI Program RAI responses sent by letter dated September 28, 2012.

The project team reviewed the plant-specific, proprietary information associated with RVI Program responses to RAIs 6 and 9.

Audit Results

<u>RAI 6</u>

Part 3 of RAI 6 stated, "If there are any components at IP2 or IP3 not bounded by assumptions regarding neutron fluence, temperature, stress or material used in the development of MRP-227-A, describe how the differences were addressed in the plant-specific RVI Inspection Plan. The staff requests that the applicant, as a part of its demonstration, discuss whether there would be any changes to the screening, categorization, failure modes, effects and consequences analysis (FMECA) process and functionality analyses if the plant-specific variables (the neutron fluence, temperature, stress values, plant-specific operating experience, and materials) are used. This evaluation should address whether additional aging mechanisms would become applicable to the component."

In its response to Part 3 of RAI 6, the applicant stated, in part that "[IP2 and IP3 have] a few components that were fabricated from CF8 cast austenitic stainless steel (CASS) material rather than the Type 304 stainless steel (SS) called out in MRP-191 (Ref. 1). These items along with the items that were fabricated from different but essentially equivalent materials are summarized in a Westinghouse proprietary document [Entergy reference 1]. Plant-specific details are proprietary and not typically released as part of RAI responses. If the NRC requires additional details, the calculation would be made available for the NRC to review the supporting technical basis."

During the audit, the staff reviewed the Westinghouse calculation CN-RIDA-12-48, "Indian Point Units 2 & 3 Support for Reactor Internals Aging Management – RAI Responses for Applicant/Licensee Action Items 1 and 2," in which the material differences for IP2 and IP3 were evaluated. There were several components with materials differing from the MRP-191 assumptions. Only the conduit support and thermocouple stop had degradation mechanisms not bounded by those applicable to the generic material in MRP-191.

The conduit support and thermocouple stop at IP2 and IP3 are CF-8 (cast stainless steel) rather than Type 304 stainless steel as assumed by MPR-191. A plant-specific FMECA was performed for the CF-8 components and determined that there were no degradation mechanisms of concern. The susceptibility of components to degradation increased from low to medium but the core damage likelihood associated with degradation of these components remained low. The overall risk ranking changed from 0 to 1, but the result of the FMECA was categorization of the components as no additional measures, which is the same as the result for the generic component in MRP-191.

Several other components had different materials at IP2 and IP3 than assumed in MRP-191. In these cases, the degradation mechanisms which would screen in for the components were the same or bounded by those applicable to the material assumed in MRP-191. Therefore, the original FMECA documented in MRP-191 is still bounding. These are listed in the table below.

Component	MRP-191 Material	IP2&IP3 Material	Comment
BMI Column Cruciform	CF-8	304 SS	No thermal embrittlement for 304 SS, so fewer degradation mechanisms
Locking Caps	304 SS	304L SS	No change to degradation mechanisms or susceptibility
Flux Thimble Tube Plugs	304 SS	308 SS	No change to degradation mechanisms or susceptibility
Upper and Lower Internals Fuel Alignment Pins	316 SS	304 SS	No change to degradation mechanisms or susceptibility
Lower Support Column Bolts	304 SS	316 SS	No change to degradation mechanisms or susceptibility
Radial Support Key Bolts	304 SS	316 SS	No change to degradation mechanisms or susceptibility
Thermal Shield Dowels	316 SS	304 SS	No change to degradation mechanisms or susceptibility
Mixing devices	CF-8	304 SS	No thermal embrittlement for 304 SS, so fewer degradation mechanisms
Lock Keys	316 SS	304 SS (IP3)	No change to degradation mechanisms or susceptibility

The NRC staff reviewed the information related to the conduit support and thermocouple stop in MRP-191. The corresponding component name in MRP-191 is the upper instrumentation conduit and supports - Brackets, clamps, terminal blocks, and conduit straps. In MRP-191 the components had no screened-in degradation mechanisms. A change in material to CASS such as CF-8 would cause thermal embrittlement (TE) to screen in based on the MRP-175 screening criteria. The new categorization (FMECA Class 1) of the IP2 and IP3 components is consistent with generic components in MRP-191 that have medium likelihood of failure and low likelihood of causing core damage if they were to fail. Therefore, the staff finds the plant-specific FMECA performed for the IP2 and IP3 conduit support and thermocouple stop acceptable. The staff also finds the applicant’s evaluation of the other components having materials differing from the generic material in MRP-191 to be acceptable because the staff agrees that the same or fewer aging degradation mechanism are applicable to the IP2 and IP3-specific materials, and that the applicable degradation mechanisms would not be more severe for the IP2 and IP3 –specific materials. There are no new RAIs identified as a result of the staff's audit related to RAI 6.

RAI 9

In Part 1 of RAI 9, the staff requested the applicant provide the specific acceptance criteria for spring height and/or hold down force from the IP2/IP3 licensing basis.

In its response by letter dated September 28, 2012, the applicant stated that "the acceptable criteria are based on the measured height of the spring as a function of time relative to the required hold-down force. Applicable plant loading conditions were evaluated. Time-dependent details for the hold down spring (HDS) height measurements are summarized in [Entergy reference 1]. Confirmatory actions are included if applicable after completion of the evaluation. Plant-specific details are proprietary and not typically released as part of RAI responses. If the NRC requires additional details, the calculation would be made available for the NRC to review the supporting technical basis."

During the audit, the staff reviewed calculation CN-RIDA-12-50. The spring height calculation determines the acceptable spring height at the end of life (60 years) to provide the required hold-down force. Acceptable spring heights for lower effective full power years (EFPY) are determined based on the assumption that the HDS height decreases linearly over time. An initial spring height measurement was taken during hot functional testing. [[

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The staff's audit of the HDS acceptance criteria calculation clarified the relationship of the acceptance criteria with the IP2 and IP3 design basis. Information provided during the audit justified the conservatism of assuming HDS height decreases linearly with time. There are no new RAIs identified as a result of the staff's audit related to RAI 9.

References

1. MRP-191 Revision 0, “Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR Designs,” ADAMS Accession Number ML091910130